NON-PUBLIC?: N
ACCESSION #: 8901310231
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Sequoyah, Unit 1 PAGE: 1 OF 13

DOCKET NUMBER: 05000327

TITLE: Reactor Trip Due To The Actuation Of The Turbine Generator Neutral Overvoltage Protective Relay Caused By A Phase-To-Ground Fault Internal To The Generator
EVENT DATE: 11/18/88 LER #: 88-045-01 REPORT DATE: 01/25/89

OPERATING MODE: 1 POWER LEVEL: 072

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: TB COMPONENT: GEN MANUFACTURER: W120
REPORTABLE TO NPRDS: Yes

SUPPLEMENTAL REPORT EXPECTED: NO EXPECTED SUBMISSION DATE:

ABSTRACT:

This LER is being revised in its entirety to provide the root cause and corrective action associated with this event.

On November 18, 1988, at 1004 EST, the unit 1 reactor tripped from a steady-state condition of 72-percent rated thermal power as a result of the main turbine being tripped due to the actuation of the main generator protective relay for neutral overvoltage. This relay monitors the secondary voltage of the generator neutral transformer and actuates when this voltage exceeds a setpoint of 10 volts for one second, indicating a phase-to-ground fault. Subsequent to the trip, Operations personnel responded by performing the appropriate emergency and general operating instructions to stabilize the unit. During the reactor coolant system (RCS) cooldown following the reactor trip, RCS average temperature (Tave) decreased below the no-load Tave of 547 degrees F due to an excessive mass loss from the steam generators via the steam dump control system. A shutdown margin calculation was performed following the trip which determined Technical Specifications shutdown margin limits were maintained. Testing of the main generator revealed a phase-to-ground fault internal to the generator on "C" phase stator coil T-17. The neutral winding of "B" phase, stator coil T-13, failed during testing of the main generator. The stator bars were removed and transported to Westinghouse for analysis. This analysis concluded the fault associated with T-17 was due to a breakdown of coil insulation by either ingress of water from outside the coil or leakage of a hollow, water carrying strand. T-13 had a hollow strand leakage which occurred due to corrosion, which was also observed in T-17. The main generator has been repaired, and further recurrence controls have been taken to limit the RCS cooldown. Shutdown margin was maintained throughout this event.

END OF ABSTRACT

This LER is being revised in its entirety to provide the root cause and corrective action associated with this event.

DESCRIPTION OF EVENT

On November 18, 1988, at 1004 EST, with unit 1 in mode 1 (72-percent power, 2230 psig, 570 degrees F), and unit 2 in mode 1 (55-percent power, 2235 psig, 563 degrees F), a reactor trip on unit 1 occurred as a result of the main turbine (EIIS Code TA) being tripped due to the actuation of the main generator (EIIS Code TB) protective relay 159GN, "generator 1 neutral overvoltage." This relay monitors the secondary voltage on the generator neutral transformer and actuates when this voltage exceeds 10 volts for one second, indicating a phase-to-ground fault. When actuated, 159GN initiates a signal to close the main turbine (EIIS Code SB) stop valves which provide an input to the reactor protection system (RPS) portion of the solid state protection system (SSPS, EIIS Code JC) to generate a reactor trip signal.

Prior to the event the following initial conditions existed:

1. Generator neutral overvoltage main control room (MCR) meter reading 4.2 to 4.4 volts (typical reading).

2. Reactor coolant system (RCS) average temperature (Tave) - 570 degrees F.

3. RCS pressure - 2230 psig.

4. Control rods in manual with bank D at 176 steps.

5. Steam generator levels - 44 percent.

6. Pressurizer level - 54 percent.

7. Pressurizer level controller selected to lowest reading (most conservative) channel.

8. Two steam dump control valves isolated.

At 1004 EST, an annunciator (EIIS Code IB) alarmed in the MCR (EIIS Code VI) on the electrical control board section alerting operators of an electrical system malfunction. Simultaneously, annunciators for a turbine (EIIS Code TA) shutdown and the transfer of the 6.9 kV unit boards (EIIS Code EA) from unit station service transformer to offsite power supply alarmed, along with the reactor trip "first out" indicating that the reactor had tripped on a turbine trip. The lead reactor operator (LRO) announced the reactor had tripped, reactor trip breakers had opened, and all control rods were fully inserted. The senior reactor operator (SRO) announced entering Emergency Procedure E-0, "Reactor Trip Or

Safety Injection," and immediate operator actions were initiated in accordance with the procedure. An extra MCR reactor operator was directed to assist the balance of plant (BOP) operator with the realignment of Turbine Building equipment.

One pressurizer spray valve had been opened prior to the trip to equalize the boron concentration of the pressurizer with that of the RCS (EIIS Code AB) loops; this valve was immediately closed by the LRO after the reactor trip. Steam dump control system (SDCS, EIIS Code JI) valves opened as designed when the required arming signal (reactor trip) was generated and RCS Tave was greater than the design no-load Tave of 547 degrees F.

As RCS Tave was decreasing, a main feedwater (HFW, EIIS Code SJ) isolation (FWI) occurred at 554 degrees F as designed on a reactor trip. The FWI tripped the HFW pumps and closed the MFW regulating, bypass, and isolation valves. The trip of both HFW pumps initiated an automatic auxiliary feedwater (AFW, EIIS Code BA) pump start signal. Both motor-driven AFW pumps (MDAFWP) and the turbine-driven AFW pump (TDAFWP) started and delivered greater than 440 gpm to each of the four steam generators. Upon verification by the operators that a safety injection signal had not actuated, and based on plant parameters that it should not have actuated, the operators transitioned to ES-0.1, "Reactor Trip Response," in accordance with E-0. Step one of ES-0.1 is to check Tave, which at the time was decreasing below 547 degrees F. ES-0.1 instructs that if Tave is not stable at 547 degrees F, the response is to verify: 1) steam dump valves and steam generator power operated relief valves (PORVs) closed, 2) steam generator blowdown valves closed, and 3) HFW isolated. Additionally, ES-0.1 states to monitor the cooldown and to take manual control of total AFW flow if necessary to control the cooldown. ES-0.1 further states to maintain total AFW flow greater than 440 gpm until narrow-range level is greater than 25 percent in at least one steam generator. Also, if Tave is less than 540 degrees F, the instruction directs the operator to borate 360 gallons at greater than or equal to 10 gpm. The BOP operator verified a FWI had occurred, steam generator blowdown valves were closed, steam generator PORVs were closed, and all steam dump valves indicated closed except 1-FCV-1-112 which was indicated in the throttled position. The operator also noticed the TDAFWP total flow indicator offscale high, indicating greater than 1000 gpm. All steam generator levels were less than 25 percent at this time (1007 EST), and AFW control remained in automatic. Since Tave decreased below 540 degrees F, the LRO operator initiated a boration of 360 gallons via the normal RCS makeup system (EIIS Code CB) at greater than 10 gpm. At approximately 1009 EST, the pressurizer level decreased to 17 percent due to the RCS cooldown and shrinkage, thereby automatically initiating an isolation of the RCS letdown line of the chemical and volume control system (EIIS Code CB). At approximately 1013 EST, steam generators 1 and 4 levels recovered to 25 percent, and manual control of AFW was taken to reduce AFW to these two loops. At approximately 1015 EST, the BOP operator reset the FWI signal and tripped the TDAFWP by closing the steam inlet valve.

At approximately 1016 EST, steam generators 2 and 3 levels returned to 25 percent, and manual control of AFW flow was taken for these loops.

The LRO noted that level was increasing in the pressurizer and, after consulting with the SRO, selected a higher indicating level channel as the controlling channel which allowed letdown to be reestablished at approximately 1016 EST. At approximately 1017 EST, the volume control tank (VCT) outlet isolation valves automatically closed due to level decreasing to seven percent in the VCT.

The refueling water storage tank (RWST) suction valves to the centrifugal charging pumps (CCPs) opened when the VCT outlet valves closed providing suction from the RWST, which contained water at a boron concentration of approximately 2000 ppm. At approximately 1020 EST, a shutdown margin (SDM) calculation in accordance with Surveillance Instruction (SI)-38, "Shutdown Margin," was completed which determined a boron concentration of 1172.5 ppm was required to meet Technical Specification SDM limits. The RCS boron concentration prior to the trip was 1247 ppm. RCS Tave reached a minimum value of 522.5 degrees F at 1030 EST.

At approximately 1038, VCT level recovered to 13 percent, allowing a manual realignment of the CCP suction from the RWST to the VCT which was performed. Also at this time, the boration of the 360 gallons was completed. After ensuring that stable plant conditions could be maintained in accordance with ES-0.1, and initial investigations indicated a main generator malfunction, Operations entered General Operating Instruction (GOI)-3, "Plant Shutdown From Minimum Load To Cold Shutdown," to reduce the unit's mode of operation to facilitate investigations and repairs of the main generator.

CAUSE OF EVENT

The immediate cause of the reactor trip was the result of the four main turbine stop valves fully closing coincident with SSPS permissive P-9 (excore power range monitors detecting greater than or equal to 50-percent power). The signal initiating the closure of the main turbine stop valves was generated from the actuation of the main generator neutral overvoltage relay 159GN. This relay actuated when the secondary voltage exceeded 10 volts for one second.

Immediately following the reactor trip, a posttrip review team consisting of multidiscipline plant personnel was assembled in accordance with Administrative Instruction (AI)-18.78, "Post Trip Review Report," to investigate and report on the root cause of the event. Additionally, a task force was assembled with a charter to investigate, and determine in conjunction with Westinghouse, the root cause of the main generator ground. This investigation was to include a review of the generator operation since initial synchronization. A posttrip review report addressing the cause of the reactor trip and an analysis of the unit performance posttrip was approved on November 30, 1988.

Subsequent to the trip and unit stabilization, TVA personnel initiated an investigation to determine the root cause of the main generator fault. TVA personnel measured the insulation resistance from the main generator to the main transformers by use of a megger test. Additional megger testing of segregated portions of the circuit concluded that a ground fault existed internal to the main generator.

Megger testing of the stator revealed that there was an insulation breakdown on "C" phase of the stator winding. DC voltage drop testing identified the point of failure in the top stator winding coil in slot 17 (T-17) located 9-1/4 inches from the turbine end. At this time, the generator was disassembled to facilitate a visual inspection of the stator windings. A visual examination confirmed T-17 to be the faulted coil. Onsite testing of T-17 by TVA and Westinghouse began by pressure testing the coil for pressure boundary integrity (evidence of stator water channel leakage from inside the coil). No leakage was detected using 100 psi nitrogen gas pressure. Insulation was isolated on the coil in 20-inch sections in preparation of electrically testing the insulation to the point of the fault. However, during this process the hollow copper conductors of the top outside layer of this coil in the arm area on the turbine end were accidentally cut with a hacksaw by the technician performing the process, thus preventing further pressure testing. At this point, a decision was made to ship the coil to Westinghouse laboratories in Casselberry, Florida, for further analysis and testing. The replacement coil for slot 17 was successfully tested and placed in slot 17, but left electrically disconnected. The stator water system was returned to service to facilitate AC high-potential (hi-pot) testing of the remainder of the main generator winding on November 28, 1988. "A" phase and "C" phase passed the hi-pot test for one minute at 32.66-kV AC. The hi-pot test was then performed on "B" phase at 32.66-kV AC and, at 26 seconds into the test, stator coil T-13 on the neutral winding failed, Coil T-13 was pressure tested with 100 psi of nitrogen pressure over a 26-minute period, prior to removal from the unit 1 main generator, to establish a degree of pressure boundary integrity. The results of this test yielded no significant leakage from stator coil T-13. A freon gas test of T-13 was attempted using a halogen detector; however, background halogen levels were too high to allow this method of testing to be successful. No further onsite testing of T-13 was performed, and the coil was then shipped to the Westinghouse laboratory for further testing and analysis. After removal of the faulted T-13 coil from "B" phase, the remainder of the "B" phase winding was hi-pot tested at 32.66-kV AC for one minute with acceptable results. The replacement coil for T-13 was successfully tested before placement into the slot.

TVA representatives along with a representative from National Electric Coil, who was contracted to perform a independent assessment of TVA and Westinghouse efforts, were dispatched to the Westinghouse laboratory.

Various electrical, physical, chemical, and metallurgical tests were performed at Westinghouse Casselberry laboratories. These tests indicate the root cause of the failure of coils T-17 and T-13 was due to breakdown of the coil insulation by the effect of water. A source of water was found in coil T-13, which was a through-wall corrosion pit in a water carrying strand. No leaking strand was located on coil T-17.

Since oil contamination was observed in the insulation on coil T-17, ingress of water with oil from outside the coil is a credible source of moisture. All evidence indicates, however, that moisture contamination lead to the failure but does not clearly identify the source of the moisture is from internal leakage of the coil or ingress of moisture from external sources. No significant amounts of hostile chemical species were detected, nor has evidence of an insulation deficiency been discovered.

In parallel to the stator testing, stator water system water and gas samples were collected for analysis. The results of the water samples indicated the system chemistry to be within limits. The stator water system gas sample results indicated evidence of an internal fault by the presence of a small amount of carbon monoxide.

The posttrip cooldown occurred as a result of an excessive mass loss from the steam generators via the steam dump control system. The RCS overcooling is further compounded by the refilling of the steam generators with water from the condensate storage tank, which is typically 70 degrees F, via the AFW system. The AFW system is designed with an automatic steam generator level control system. AFW to each steam generator is controlled to automatically maintain a no-load level of 33 percent. On a FWI, two, 100-percent-capacity MDAFWPs and one, 200-percent capacity TDAFWP start and deliver full flow until steam generator levels are returned to 33-percent narrow-range level.

ES-0.1 had been revised as a result of a review of unit 2 trip sDm data to address manual AFW control by incorporating the Westinghouse Owners Group (WOG) guidelines applicable to the emergency operating procedure (EOP). These guidelines, initially incorporated into Revision 3, removed in Revision 4, and readded in Revision 5, included instructing the operator to maintain total AFW flow greater than 440 gpm until narrow-range level is greater than 25 percent in at least one steam generator (S/G). This guideline was not specific enough to change a previously ingrained operator mindset. (With an automatic AFW system and the cooldown issue not yet identified, Sequoyah licensed operators had been previously trained that an indication of maximum AFW flow is appropriate when verifying AFW flow after a reactor trip.) The intent of this revision was to limit total AFW by taking manual control immediately after the trip to an amount slightly higher than 440 gpm until the level in at least one S/G was greater than 25 percent. This intent was not met during the November 18, 1988 unit 1 trip due to a philosophy of not interrupting automatic safety system functions

of the plant being ingrained in the operators during their entire training program. The operators mindset was to allow full AFW flow posttrip and take manual control of AFW to individual S/G s to limit flow only after levels recovered. By allowing the system to remain in automatic control following the November 18, 1988 trip until the S/G levels recovered, excessive amounts of cool AFW were delivered to the S/Gs, and the undesired cooldown of the RCS was not prevented.

Possibly contributing causes of the ineffective correction action following the unit 2 trips includes inadequate reinforcement following initial training on manual AFW control and lack of accuracy in simulator modeling.

The cause of the VCT level decreasing to below seven percent was a combination of the chemical and volume control system letdown line isolating on low pressurizer level due to the RCS cooldown and normal VCT makeup being reduced from approximately 75 gpm to 10 gpm when the makeup mode select switch was selected to BORATE to comply with ES-0.1.

The cause of the total TDAFWP flow indicator reading offscale high was the result of the flow controller field tuner being out of calibration.

The cause of 1-FCV-1-112 indicating in the throttled position subsequent to the reactor trip was due to a misaligned closed limit switch. It has been determined that the valve was closed when the throttled position was observed by the MCR operator.

ANALYSIS OF EVENT

This event is being reported in accordance with 10 CFR 50.73, paragraph a.2.iv, as an event which resulted in the automatic actuation of Engineered Safety Features (ESF). A notification by telephone was made to the Nuclear Regulatory Commission within four hours from the event in accordance with 10 CFR 50.72, paragraph b.2.ii.

Following the trip, a RCS cooldown below no-load Tave occurred, as has been previously discussed. A SDM calculation was completed using a conservative Tave value of 510 degrees F compared to the actual minimum RCS Tave of 522.5 degrees F. A boron concentration for a shutdown reactivity of 1600 ppm, assuming the most reactive rod stuck out, was calculated to be 1172.5 ppm. The RCS boron concentration prior to the trip was 1247 ppm.

This represents a safety margin of 74.5 ppm without any additional boration. Therefore, Technical Specifications (TS) and Final Safety Analysis Report limits were not challenged with respect to the capability to mitigate the consequences of a concurrent steam line break at the time of RCS cooldown.

The safety related SSPS logic performed as designed to mitigate the consequences of the main turbine trip by initiating a reactor trip. Therefore, this event did not impact the health and safety of plant personnel or the general public.

CORRECTIVE ACTION

Immediate corrective actions were to bring the plant to a stable condition in accordance with E-0, ES-0.1, and GOI-3.

A night order was issued to shift operating personnel describing the unit I event and the operator actions required to comply with ES-0.1.

The remainder of this section will be divided into two portions to address separately the subsequent corrective actions associated with the generator fault and the RCS cooldown condition.

GENERATOR FAULT

The following corrective actions have been completed:

1. New stator coils were tested and installed in slots 13 and 17.

2. Seal oil coolers and hydrogen coolers were leak tested satisfactorily.

3. Generator internals were cleaned of oil prior to reassembling.

4. Portions of the Westinghouse generator aids (GenAids) enhancement package were installed.

(a) Dual tower gas dryers to increase hydrogen drying efficiency.

(b) Radio frequency noise detector to monitor the generator windings for impending failures.

5. Overcurrent signal sensing relays that monitor the difference in potential to ground were installed in the relay room to protect entire winding during operation.

6. Appropriate operations procedures were revised to incorporate the generator modifications that impacted the operation of the cooling systems.

All testing recommended by Westinghouse to provide assurance of suitability for operation has been performed, and diagnostic testing indicated the generator suitable for service. Unit 1 presently is at 100-percent power with no problems relevant to the cause of the previous trip identified with the main generator. The degraded condition of the faulted coils was not and could have not been identified during a visual inspection of the generator winding during the extended Mode 5 outage.

The following enhancements are being considered by TVA as follows:

1. Complete installation of the Westinghouse GenAids package for both units.

2. During unit shutdown periods, the generator will be "layed-up" in accordance with system procedures.

3. TVA is considering rewinding both units' main generator sometime in the future.

RCS COOLDOWN

TVA initially identified the posttrip RCS cooldown below design no-load Tave on June 14, 1988 during a review of SDM data recorded subsequent to the unit 2 reactor trips after startup from the extended Mode 5 outage. This condition was documented in LER SQRO-50-328/88030, and TVA committed to provide details of an action plan to resolve the SDM issue by September 15, 1988.

An August 31, 1988 TVA submittal to the NRC included the results of a Westinghouse analysis on TS SDM requirements and stated that ES-0.1 had been revised to require a manual boration if Tave decreased below no-load Tave. This revision also required the operations crew to initiate manual control of AFW if RCS Tave decreased below 530 degrees F. This submittal contained a commitment to provide a status on determining appropriate resolution to the cooldown issue by October 14, 1988. Revision 1 to LER 88030 was submitted on September 9, 1988 and stated ES-0.1 was revised (Revision 4, July 16, 1988) to provide operator action to initiate a boration based on Tave, to provide adequate SDM during the cooldown after the reactor trip, and deleted the requirement for manual control of AFW. This submittal stated that the long-term resolution to the cooldown issue would be made in the followup report to the August 31, 1988 letter to the NRC.

An October 5, 1988 TVA letter provided a list of 10 commitments made by TVA in a NRC Management Meeting regarding SDM on September 13, 1988, which included manual control of AFW. An October 14, 1988 TVA letter provided the NRC a description of the root cause of the posttrip cooldown condition and the corrective action to prevent recurrence of this condition: 1) ES-0.1 would be revised, prior to the restart of unit 1, to address manual AFW control and change the boration value to bound both units 1 and 2 and 2) the steam dump controls will be modified, no later than February 28, 1989, to prevent excessive blowdown of the S/Gs following a reactor trip.

ES-0.1 was revised (Revision 5, October 20, 1988) to state that if a cooldown continues after verifying steam dump valves, S/G PORVs, S/G blowdown valves are closed and MFW isolated, and if Tave is less than 547 degrees F, the operator is to maintain total AFW flow greater than 440 gpm until narrow-range level is greater than 25 percent in at least one S/G. Revision 5 also changed the boration value from 322 to 360 gallons at greater than or equal to 10 gpm if Tave is less than 540 degrees F to bound units 1 and 2.

Unit 1 reactor tripped on November 18, 1988 as described earlier in this report. The corrective action described and taken as a result of the unit 2 posttrip cooldowns was ineffective in that TVA had not taken adequate measures to change the licensed operators mindset on previously ingrained posttrip actions. Specifically, ES-0.1, Revision 5, did not contain adequate detail on manual AFW control, and subsequent operator training was not adequately reinforced to change this mindset.

Subsequently, ES-0.1 was clarified and presently (Revision 8) reads as follows:

Step one of ES-0.1, which is applicable to both unit 1 and unit 2, is to cheek Tave and if Tave is less than 547 degrees F, then

1. Verify steam dumps and S/G PORVs closed.

2. Verify auxiliary steam header supplied from other unit or auxiliary boiler.

If cooldown continues, then manually control total AFW flow in accordance with E-FOP, "Foldout Page." Maintain total AFW flow to all S/Gs to a minimum of 440 gpm (approximately equal to 100 gpm to each S/G) and do not exceed 500 gpm until at least one S/G is greater than 25-percent narrow range.

When Tave is greater than or equal to 547 degrees F, then increase AFW flow to restore S/G level.

When level is greater than 25 percent in at least one S/G, then manually control total AFW to minimize RCS cooldown.

When Tave is less than or equal to 540 degrees F, then emergency borate 360 gallons at greater than or equal to 75 gpm of greater than or equal 20,000 ppm boron using FCV-62-138 in accordance with Abnormal Operating Instruction (AOI)-34," Emergency Boration," and if the cooldown continues due to steam leakage and Tave is less than 530 degrees F, then close main steam isolation and bypass valves.

If Tave is greater than 540 degrees F or SDM verified with RCS temperature increasing, then boration can be terminated.

If Tave is less than 500 degrees F, then emergency borate at greater than or equal to 75 gpm of greater than or equal to 20,000 ppm boron using FCV-62-138 in accordance with AOI-34 until one of the following criteria is met:

1. SDM verified according to SI-38.

2. Tave stable or increasing.

3. Total boration is greater than 2500 gallons of greater than or equal to 20,000 ppm boron.

The Sequoyah simulator software was reprogrammed to more accurately model the actual plant cooldown. Simulator training on Revision 8 to ES-0.1 to licensed shift operators was completed on December 9, 1988. This training included available boration flow paths to meet the boration requirement within the instruction. Additionally, a TVA training evaluation was conducted on December 19, 1988 to determine the effectiveness of the ES-0.1 training. One operating shift crew was notified after shift turnover to report to the training center with the exact nature of the training not made available to them. The crew was subjected to a reactor trip simulator scenario and subsequent questioning on ES-0.1 procedure bases. The evaluation concluded that the crew performed satisfactorily and would have successfully mitigated a posttrip cooldown if this had been an actual reactor trip condition.

Work Request (WR)-B283483 was initiated to investigate and repair the cause of the TDAFWP total flow indicator to read greater than 1000 gpm instead of 880 gpm during the event. During troubleshooting of the circuit, the flow controller, FIC-46-57, field gain was identified out of calibration and not limiting flow to 880 gpm. The controller was field tuned and tested to limit flow to 880 gpm.

WR-B280330 was initiated to investigate and repair steam dump control valve 1-FCV-1-112 which was observed in the MCR to be in a throttled position with the remaining valves in the closed position subsequent to the reactor trip. The closed limit switch was found slightly out of adjustment. Open and closed limit switches were adjusted and tested for 1-FCV-1-112.

As a result of detailed analysis of the posttrip cooldown condition, the following hardware modification was generated. Design Change Notice (DCN)-M968A has been incorporated in unit 1 which reduced the low Tave FWI setpoint from 554 to 550 degrees F which will result in an increased posttrip RCS temperature by extending the period of time MFW is allowed to

be delivered to the S/Gs. This DCN also increased the reactor trip controller setpoint in the SDCS from 547 to 552 degrees F which will result in less mass lost from the S/Gs following a reactor trip. DCN-M969A, applicable to unit 2, is scheduled to be implemented prior to startup from the next refueling outage.

TVA has determined that the corrective actions enumerated below will be taken to prevent and/or further mitigate the posttrip cooldown at Sequoyah:

1. Training to licensed shift operators on the steam dump control modification, and the FWI setpoint change will be included in the 1989 requalification training program.

2. A posttrip cooldown scenario will be incorporated into the annual requalification training program. The shift operating crews will be formally evaluated during week one 1989 requalification training on their actions during this scenario.

3. DCN-M969A, unit 2 SDCS reactor trip controller modification, will be implemented prior to startup from the cycle 3 refueling outage.

The following additional actions will be taken to address generic implications of ineffective corrective actions previously taken for the cooldown condition:

1. TVA will assess management involvement in the posttrip/incident review process and associated corrective action, This will be completed by March 31, 1989.

2. TVA is presently conducting a review of LERs resulting from ESF actuations including reactor trips for 1988 to identify common root causes and/or insufficient corrective action. This project will be completed by March 31, 1989.

3. An Operations training instructor will observe MCR activity on a monthly basis to evaluate licensed and requalification operator training effectiveness. This action will continue until results indicate no further action is required.

4. The simulator modeling will be verified before implementation of any future posttrip/incident investigation identified corrective actions requiring simulator training.

ADDITIONAL INFORMATION

Detailed in this report is the first reactor trip with the reactor trip breakers closed since the unit 1 startup from an extended shutdown of approximately three years (August 1985-November 1988).

TEXT